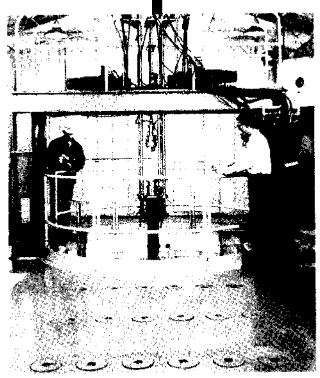
Water reactor safety research aids in understanding of reactor behavior

by Rita Scott, EG&G Idaho

1955

Atomic Energy Commission begins water reactor research at INEL. SPERT I begins testing.



THIS 1961 PHOTO SHOWS SPERT I as it was—an open tank vessel with control rods in the upper structure. The reactor was housed in a tin building.

1962

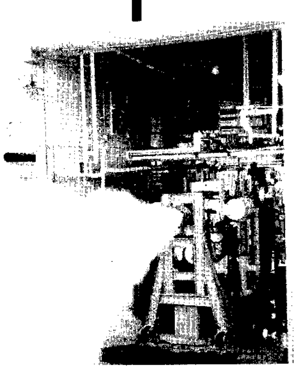
SPERT I destructive test.

1965

Construction began on Power Burst Facility.

1968

First modification of Semiscale making the single loop.



THERE WAS NO suppression tank on this early model of Semiscale, the Single Loop Semiscale Blowdown Test Facility. The coolant was released into the atmosphere. This photo was taken in 1968.

When the Atomic Energy Commission began investigating the safety of water-cooled reactors at the Idaho National Engineering Laboratory (INEL) in 1955, the research was focused on the major safety concern of that time, the "runaway power" or reactivity accident. Researchers wanted to understand reactor behavior and the consequences of such an accident, caused by excessive reactivity which produces a deviation from the desired chain reaction. The question asked was, "Would the system shut itself down because of the mechanisms built into the reactor before it destroyed itself?"

The Special Power Excursion Reactor Test (SPERT) program was set up to answer the question. In 1955, SPERT I, an open pool reactor in an unshielded tank, began testing in a tin building. During the next 10 years, three other SPERT reactors and a central control station were built in an arc one-half mile from each other. At one time, all four reactors were operating at the same time from four control rooms in that station.

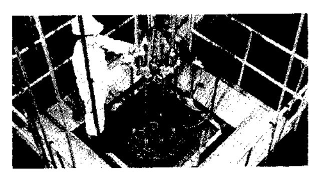
These small reactors originally used plate-type cores that could be pushed far beyond the limits of safe reactor operation as it was then understood. They were also designed so that a wide range of variables such as plate design, core configuration, coolant flow, temperature and pressure coefficients could be studied.

According to Clyde Toole, Operations Manager of the SPERT program, the purpose of SPERT was to study the kinetic behavior of the reactor during off-normal conditions. The SPERT cores were unique in that they contained a transient rod which was used to pull poison (neutron absorbers) into the core. Reactivity was then added by firing the transient rods out of the core, causing the power to rise.

Eleven different cores were tested in SPERT I before a destructive test was conducted. Toole says this test was carefully planned. "We waited for the weather to be just right, the roof was taken off the building, and radiation monitors were set up all around the site."

Then a series of tests were run using increasing amounts of excess reactivity until the core destructed, causing a steam explosion. "Water shot 80 feet in the air," Toole recalls.

But contrary to site folklore, this was not the last of SPERT IV. According to Toole, a different type core, one with round fuel rods of uranium oxide from the N.S. Savannah Critical Facility was tested at SPERT I after the plate-type fuel rod core was destroyed. Destructive tests were attempted on this core also. However, because it proved to be much more resistant than the earlier core, it was put to use in another SPERT reactor, SPERT IV.



PICTURED IS THE SPERT IV core in its later configuration and design. This rod type core is known as the Capsule Drive Core. Round fuel rods were used -the long rods being the control rods for the reactor. The Capsule Drive Core was the forerunner of the present day Power Burst Facility. The operator shown in the photo is Vic Kelsey, presently employed at PBF.

Other SPERT reactor experiments were less dramatic. SPERT II was developed to use heavy water as a core moderator and/or reflector; SPERT III was a very high temperature and pressure reactor which could run at conditions similar to power reactors; and SPERT IV was designed to investigate reactor stability.

Toole says the N.S. Savannah Critical Facility core was put in SPERT IV in such a way that a cavity was left in the center so that a capsule could be inserted into it. This core was run at very short periods at very high power and it did not damage itself, but did cause failure of the test samples in the capsule. These were called the Capsule Drive Core tests.

Power Burst Facility (PBF)

PBF was the outgrowth of these SPERT IV tests. In the SPERT reactor, however, the tests within the capsule were conducted in a static condition, with no water flowing or pressure or temperature variance. PBF added the capability to control the environment in the test space. Construction began on the facility in 1965 and it achieved criticality in 1972.

PBF, which is near the site of the SPERT I reactor, has an open tank reactor vessel; the control room is one-half mile away.

The water reactor safety research at INEL, which includes the Thermal Fuels Behavior Program, is conducted by EG&G Idaho at PBF for the Department of Energy. This program is responsible for performing light water reactor fuel behavior studies as part of the Nuclear Regulatory Commission's Water Reactor Safety Research Program. The Thermal Fuels Behavior Program also conducts in-pile testing of instrumented fuel assemblies in the Halden Reactor in Norway.

PBF is designed to produce intense power bursts capable of melting test fuel samples without damage to the facility itself. All tests are high priority, selected to get information on the behavior of fuel rods under a wide variety of operating conditions and during hypothetical accident sequences—abnormal reactor conditions.

The facility has an in-pile test loop, much like a giant test tube, into which single and clusters of fuel rods are put. This test loop has its own environment, separate from the core. Operators can control the test fuel rod coolant flow rate, temperature and pressure so that they are typical of a pressurized water reactor or a boiling water reactor at hot-standby condition. With the use of high-speed valves, the in-pile tube portion of the loop can be rapidly depressurized in a way that is similar to a loss-of-coolant accident in a light water reactor.

PBF can be operated at three levels of power intensity: 28 megawatts indefinitely, 1350 megawatts for a few minutes, and a peak power of 270 gigawatts for two thousandths of a second.

According to Cal Doucette, PBF facility manager, the original mission of PBF was a 40 test series program. Over the years, additions and deletions have been made to this original series of tests, including modification of the facility to permit a loss-of-coolant accident series. But with the completion of the TC-4 test, scheduled for May 1981, Doucette says the original 40 test series will be completed. Future tests will consist of operational transient tests and the severe fuel damage series.

Semiscale

A small facility at INEL's north end, then called 1/4-scale LOFT, went into operation in the early 1960s as part of the Atomic Energy Commission's Separate Effects Program. Plans called for PBF data to be compiled with information received from 1/4-scale LOFT tests and used in LOFT-U, a program designed to study the effects of a reactor core meltdown and subsequent release of fission products to the containment.

1/4-scale LOFT, so called because it was a direct adjunct to the LOFT-U program, later became known as Semiscale, a non-nuclear test program that has undergone a number of modifications since those early days of water reactor safety research.

Originally modeled after the LOFT system, Semiscale was a simple simulated reactor vessel with some pipes. Loss-of-coolant tests were conducted by expelling coolant through a blowdown nozzle.

The first modification took place in late 1968. This made Semiscale a system with a single circulating primary coolant loop with the capability of simulating various pipe breaks and included a simulated reactor core of 120, nine-inch electrically heated rods with a power capability of 1 megawatt. An emergency core coolant injection system was later added.

According to Danny Olson, a former manager of Semiscale, not a lot of attention was paid to the system's scale at that point. "The system was put together piece by piece," Olson says. It was designed to conduct experiments that could be compared with computer code predictions, not necessarily to represent phenomena in a reactor.

"We were boot-strapping—doing an experiment, checking it with the computer codes to see if it described or predicted the experiment results, and then modifying the experiment or the codes. The results of these tests helped to provide confidence in the codes that were being used to design LOFT."

Some of the early Semiscale tests were conducted to see how much water remained in the reactor vessel after a blowdown—emergency core cooling sequence. "The tests showed there was very little," Olson says. The facility did not have a suppression tank to catch the water and it was released to the atmosphere. Red dye was added to the emergency core coolant and motion pictures were taken of the fluid discharged from the system during blowdown. Olson says that when pink water came out, it was correlated with the time that the emergency core coolant injection was started. With this correlation came the knowledge that emergency core coolant was bypassing the core and going directly out of the break. This occurred because the design of the original vessel was not typical of reactor vessel and the discovery led directly to the emphasis on scaled system designs for future experimental work.

The 1-1/2 loop modification began in early 1972 when the test system was dismantled and a core, simulated with 66-inch electrically heated rods, was added as well as a complete operating loop and a passive loop where breaks could be simulated by opening rupture disks.

After the core simulator burned out during systems testing, the modification plans were expanded. Mod-1 went on the drawing boards. It was designed to investigate the effects of physical scale (by close correlation with LOFT results) and how the issue of physical scale related to the understanding of loss-of-coolant accidents and the actions of the emergency core cooling systems.

While Mod-1 was being prepared, the 1-1/2 loop system was operated in a series of separate effects tests that gave information on two-phase flow (steam-water mixture), pump behavior and critical flow rates. A transparent plexiglass vessel was put to use during this time, which investigated the countercurrent flow behavior and the reflood of a core after a break.

"Mod-1, which operated between 1974 and 1977, was the first real attempt to provide scale to the system," explains Olson. "This was accomplished by adding an active steam generator to the operating loop and pump simulators to the broken loop."

Mod-1 was scaled primarily to LOFT (1/30) rather than a pressurized water reactor. Until LOFT began operating in 1978 (LOFT is 1/50 the size of a pressurized water reactor), Semiscale provided the only information on integral systems performance that the Nuclear Regulatory Commission received.

In 1978 a 12-foot heated core, and a complete active broken loop with a scaled pump, piping and steam generator scaled primarily to a commercial pressurized water reactor rather than LOFT, was added to the system, designated Mod-3 Semiscale.

During the Three Mile Island accident, the facility was called upon to provide information quickly on the hydrogen bubble forming in the containment vessel of Unit 2, and subsequently to reproduce the entire sequence of events of the first two or three hours of the TMI accident.

After TMI, the emphasis in water reactor safety shifted to small break tests. This emphasis brought about another modification of Semiscale, Mod-2A, which was completed in 1980.

According to Paul North, manager of the Water Reactor Research Test Facilities Division, this substituted a 38-foot, full-height steam generator for the one scaled to LOFT. North says that for small break tests, full height was important. This modification brought the facility as closely as possible to a commercial pressurized water reactor power plant, in terms of thermal-hydraulic behavior.

However, the relationship between LOFT and Semiscale will continue. The latest modification will allow the facility to provide test results that evaluate the subtle differences in behavior resulting from differences in the components' physical size.

"If we run the same test LOFT is running, and our tests predict what is happening in LOFT, and it is proven, then the LOFT results can be projected with more confidence to a commercial power plant," North explains. "We can save LOFT from running a test if we have learned what we need from a Semiscale test."

LOFT

Another facility at the north end of the INEL site has also undergone a considerable number of modifications and design changes since the project was authorized in 1963. The Loss-of-Fluid Test (LOFT) facility has earned international recognition as the only nuclear facility with the capability to conduct simulated loss-of-coolant accidents. The story of LOFT and how it grew from a one-time test reactor to the current complex, internationally acclaimed facility, will be told in the next issue of the INEL News.

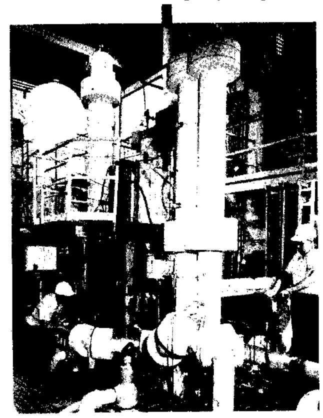
1981
40 test series completed at Power Burst Facility.

1980
Mod-2A added full size steam generators in both loops.

1979
Three Mile Island assistance provided by Semiscale.

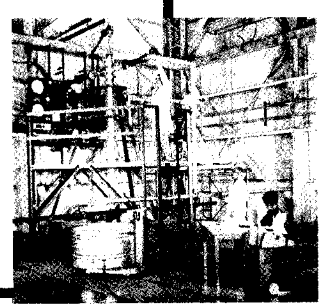
1978
Mod-3 Semiscale began scaling to commercial water reactor components.

1974
Mod-1 Semiscale began operating.



THE MOD 1 Semiscale System was the first attempt to actually scale the facility to LOFT.

1972
Power Burst Facility achieved criticality.
Semiscale 1-1/2-loop modification begun.



THE TRANSPARENT (plexiglass) vessel used for countercurrent flow tests was part of the AEC's Separate Effects Tests in 1972. This program made use of Semiscale after the core simulator had burned out and Mod 1 was being built.

1970
Power Burst Facility construction completed.